Attachment 2

TECHNICAL WORK TO SUPPORT RULEMAKING FOR A RISK-INFORMED ALTERNATIVE TO 10 CFR 50.46, APPENDIX K AND GDC 35

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1. INTRODUCTION

In Attachment 1, a feasibility assessment was performed regarding risk-informing the technical requirements of 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” and related requirements in Appendix K to 10 CFR Part 50, “ECCS Evaluation Models,” and General Design Criterion (GDC) 35, “Emergency core cooling.” The assessment was performed to existing requirements in four major areas:

1. Acceptable evaluation models for calculating emergency core cooling system (ECCS) performance
2. Acceptance criteria for ECCS performance following postulated loss-of-coolant accidents (LOCAs)
3. Assurance that the ECCS function can be accomplished reliably
4. The spectrum of breaks for which ECCS cooling performance must be calculated

The staff concluded that changes to the first three parts above of 50.46 are justified and the last is potentially feasible. Consequently, the staff:

- recommends changes to the technical requirements of the current 10 CFR 50.46,
- recommends development of a risk-informed alternative to 50.46, and
- plans on continuing the feasibility assessment regarding the spectrum of breaks.

Each of these activities, however, requires further technical work to either support a proposed rulemaking or determine the feasibility. This additional technical work, for each of the above, is discussed in detail in this attachment.

2. Description of Changes to Current 50.46 and Associated Technical Work

This section provides a description of the various proposed changes to the current10 CFR 50.46, Appendix K and GDC 35. Each section provides a description of the proposed changes and a summary of the technical work that is needed to support the rulemaking effort.

2.1 ECCS Evaluation Models

2.1.1 Proposed Change

Two options for ECCS evaluation models are proposed: the first option involves an update to the Appendix K model, in which unnecessary conservatisms are removed; the second option involves, unchanged, the existing realistic (i.e., best-estimate) model with propagation of uncertainties.

In the first option, the changes to Appendix K would permit the use of a more realistic alternative to existing decay heat requirements of Appendix K to 10 CFR Part 50. Specifically, the changes would permit use of the 1994 decay-heat standard with an NRC-approved method to account for uncertainty. Also, new information will be evaluated to determine if the following changes can be made:

- Delete the requirement involving pressurized water reactor (PWR) reflood steam cooling limitations,
- Replace the existing Baker-Just zirconium steam model with the Cathcart-Pawel zirconium steam oxidation model for heat generation, and
- Modify the requirement which prohibits return to nucleate boiling during blowdown, to allow return to nucleate boiling if justified.

In the first option, changes to Appendix K, licensees would be expected to remain vigilant in assuring that the required features of Appendix K continue to compensate for known nonconservatisms of Appendix K evaluation models.

Either of the options could be applied to a particular class of design-basis LOCAs. For example, the second option, best-estimate analyses with uncertainty propagation could, as in current practice, be applied to analyze the largest LOCAs while the first option, Appendix K models with unnecessary conservatisms removed, could be applied to all other LOCAs.

2.1.2 Needed Technical Work

The previous section described possible changes to the Appendix K ECCS evaluation model to remove unnecessary conservatisms. Many of the "required and acceptable features" of Appendix K models are specified to ensure that certain phenomena are considered in the analysis. Other features require sensitivity analysis to assure that a conservative, or at least a realistic selection is made for the subject parameter. Requirements specifying analyzed power level and peaking factors (paragraph I.A) are the most important boundary conditions in the ECCS analysis. It is not proposed these requirements be changed.

Only a few requirements actually contain specific conservatisms that could be considered for change. The U.S. Nuclear Regulatory Commission (NRC) staff has identified four requirements in Appendix K that warrant consideration for change. They are:

1. Decay Heat (paragraph I.A.3 & 4)
2. Heat Removal by ECCS (paragraph I.D.3-6)
3. Metal Water Reaction (paragraph I.A.5)
4. Nucleate Boiling Return (paragraph I.C.4.e)

Decay Heat (I.A.3 & 4) - Appendix K currently requires decay heat be modeled using the draft 1971 American Nuclear Society (ANS) standard for decay heat with a multiplier of 1.2. Information derived over the last 25 years would permit the use of a newer decay heat standard, which involves more sophisticated uncertainty methods and a greater number of model parameters and options. Sensitivity studies have indicated that a newer standard could reduce predicted peak cladding temperature (PCT) by as much as several hundred degrees. The 1994 ANS standard considers much more recent available data and methods. The 2s uncertainty as evaluated in the new standard is significantly less than the 0.2 value described in the 1971 standard. For any significant change in Appendix K evaluation models, such as decay heat, it should be determined that sufficient conservatism remains in the resulting analysis with respect to performance variables, in particular peak cladding temperature.

The uncertainty methods in the 1994 ANS standard will be assessed to assure that they can be reliably applied to Appendix K ECCS calculations. It is expected that the result would be a single multiplier, a time dependent multiplier or a simple uncertainty method. It should also be determined if separate uncertainties for actinides and recoverable fission energy should be included.

For Appendix K, additional analysis will be required to identify acceptable and practical model options and parameter values. This will include guidance regarding:

1. Whether the reactor operating history should be represented by a histogram of multiple irradiation intervals or must be modeled as a single interval,
2. Appropriate durations (T_a) of the irradiation interval(s),
3. Values of the recoverable energy per fission (Q_i) for U-235, Pu-239, and U-238, and Pu-241 (the contribution of each of these nuclides to decay heat is inversely proportional to its recoverable energy per fission),
4. The correction factor G(t) for neutron capture in fission products, and
5. The value of the atoms of U-239 produced per second per fission R, which is used in the model of U-239 and Np-239 decay heat power.

Heat Removal by ECCS (I.D.3-6) - Heat removal by ECCS includes several phenomena that are part of the heat removal process (coolant void generation, entrainment, reflood heat transfer, and certain system effects). The PWR reflood steam cooling limitations (I.D.5.b) leads to non-physical, computationally difficult and often unnecessarily conservative modeling requirements. Data developed since promulgation of 50.46 in 1974, could support deletion of this requirement. Evaluation of this data would be required to determine its applicability to the spectrum of PWR large break reflood analyses.

In small-break LOCAs (SBLOCAs) heat removal is dominated by system effects (e.g., natural circulation, steam generator heat transfer, ECCS pump capacity, operator actions). It should be noted that none of the specific requirements of Section I.D. of Appendix K, “Post-Blowdown Phenomena; Heat Removal by ECCS” are directly related to small breaks.

Metal-Water Reaction (I.A.5) - Metal water reaction can be an important heat source during periods of extended core uncovery, when the cladding reaches elevated temperatures. Appendix K currently requires the use of the Baker-Just zirconium steam model, which is known to be conservative compared to more recent data and models (e.g., the Cathcart-Pawel zirconium steam oxidation model). Previous studies showed that the effect in terms of peak cladding temperature is less than 80°F compared to less conservative models, when the peak cladding temperature remains below 2200°F. This is obviously less than the several hundred degrees for decay heat. If a revised acceptance criteria allowed peak cladding temperatures to exceed 2200°F, the effect could be very large depending on temperature. This is due to the exponential temperature dependence for the rate of this exothermic reaction. Also, metal water reaction rate is currently tied to the 2200°F PCT and 17% equivalent cladding reacted (ECR) embrittlement criteria discussed in Section 2.3, which discusses post-quench ductility as an alternate criterion. Changing the metal water reaction model would only apply to the heat release associated with oxidation. Models affecting embrittlement are described in Section 2.3.

Nucleate Boiling Return (I.C.4.e) - The Appendix K prohibition on return to nucleate boiling during blowdown is most important for plants that are predicted to have PCT early in the LOCA. It is not important for SBLOCA, since core design and experimentation have demonstrated that only core coverage is required for SBLOCA heat removal. To provide large-break LOCA (LBLOCA) relief from this requirement would involve a review of blowdown heat transfer data with appropriate “minimum film boiling temperature” data. The subject has not been extensively studied, and large uncertainties still exist. A review will be necessary to determine if current information is sufficient to allow relaxation of this requirement.

In addition to the technical work required to support removal of unnecessary conservatisms from the existing Appendix K ECCS model, there is a small amount of work that would be performed with respect to the existing best-estimate model. Paragraph (a)(1)(i) of 50.46 and Regulatory Guide 1.157 describe the features of a realistic (best-estimate) ECCS evaluation model including evaluation and propagation of uncertainties. There would be no fundamental change to that regulatory guidance for this alternative. However, some citation of the 1994 ANS decay heat standard could be added to Regulatory Guide 1.157 and other reference to more recent information could be added if that is deemed necessary.

A summary of the needed technical work to support rulemaking for the proposed changes associated with the ECCS evaluation models is provided in Table 2-1. This table also includes estimates of the time and resources required.

Table 2-1 Summary of Needed Technical Work (ECCS Evaluation Models)

Technical Work	Calendar Time to Perform Work	Resources Needed
• Assess uncertainty methods in 1994 ANS decay heat standard • Identify acceptable and practical decay heat model options and parameter values • Evaluate current (i.e., post-1974) reflood heat transfer data • Assure new metal water reaction models can be implemented properly • Evaluate current data on blowdown heat transfer and minimum film boiling temperature • Support modification of regulatory guide (RG) 1.157 to include 1994 ANS decay heat standard and other recent information	6-12 months	1.1 FTE $350K

2.2 ECCS Acceptance Criteria

2.2.1 Proposed Change

Two of the acceptance criteria in 10 CFR 50.46(b) were developed specifically for fuel rods with zircaloy cladding, and paragraph (a) restricts the application of 10 CFR 50.46 to zircaloy and a similar ZIRLO cladding alloy. These two criteria are the 2200^{o}F peak cladding temperature limit and the 17% total cladding oxidation limit. It would be possible to remove the restriction on the application of 10 CFR 50.46 to zircaloy and ZIRLO by replacing the numerical values in the regulation with the underlying principle on which they were based. That principle is described in the Opinion of the Commission in the matter of the ECCS rulemaking hearing [Ref. 1].

During a LOCA, fuel rod cladding would be subjected to a high temperature transient in steam and it would experience rapid oxidation. Oxidation at high temperatures leads to changes in the metallurgy of the cladding and can result in complete embrittlement. To ensure that the cladding would remain intact during a LOCA, the Commissioners concluded that ductility should be retained after recovery from the high temperature transient. The presence or absence of ductility was determined from a mechanical test on rings of zircaloy cladding material that had been subjected to a LOCA-like high temperature transient and then quenched. Using data from Oak Ridge National Laboratory, the Commission concluded that ductility would be retained if the percent of the cladding thickness that was oxidized did not exceed 17% and the temperature of oxidation did not exceed 2200^{o}F.

The ECCS acceptance criteria in the current 10 CFR 50.46 would be replaced by a performance-based requirement to demonstrate adequate post-quench cladding ductility and adequate core-coolant flow area to ensure that the core remains amenable to cooling, and for the duration of the accident, maintain the calculated core temperature at an acceptably low value and remove decay heat. Degradation to a noncoolable core geometry can only occur if the ductility criteria are exceeded, or if there is excessive ballooning of the cladding resulting in inadequate core-coolant flow area. As noted in Chapter 5 of Attachment 1, cores with more damage may be coolable, but phenomenological uncertainties and computational capabilities, as a practical matter, preclude the use of less conservative criteria. The current ECCS acceptance criterion related to global hydrogen generation is generally not controlling, and is not included in the risk-informed alternative. Global hydrogen generation is adequately dealt with by the hydrogen rule, 10 CFR 50.44.

It is anticipated that three options for assuring adequate ductility would be described in a regulatory guide. The first option would retain prescriptive acceptance criteria, though the specific criteria would now be contained in the regulatory guide, not in the actual regulation. This allows changes to be made to the prescriptive criteria, based on new information or for application to new cladding types, without necessitating another rulemaking. As a practical matter, for the time-being, the regulatory guide would specify the existing ECCS acceptance criteria for peak cladding temperature ($\leq$ 2200^{o}F) and local oxidation ($\leq$ 17% of the cladding thickness), which would only be applicable for zircaloy and ZIRLO cladding. The second option would be a performance-based option, where the peak cladding temperature and local oxidation criteria are replaced by a more general requirement that adequate post-quench ductility be demonstrated. Specifically, the staff would adopt a criterion of non-zero post-quench ductility for fuel rod cladding. This performance-based approach, which would be applicable to zircaloy, ZIRLO, or any other cladding alloy, would provide flexibility for current and future fuel designs by eliminating reference to cladding types and numerical values. Procedures for determining post-quench ductility could be described in the regulatory guide.

The third option would apply to embrittlement criteria that had previously been approved as part of an exemption from 50.46(b), which would continue to be accepted.

2.2.2 Needed Technical Work

Current NRC research at Argonne National Laboratory is exploring post-quench ductility of high-burnup zircaloy cladding, and this research could provide information to support development of a regulatory guide. Funding for the research at Argonne National Laboratory is already in the budget so additional funding would not be needed. Interpretation of test results and formulation of the technical basis for a regulatory guide would be performed by the NRC Technical Advisory Group on Fuel (TAG-F).

A summary of the needed technical work to support rulemaking for the proposed changes associated with the ECCS acceptance criteria is provided in Table 2-2. This table also includes estimates of the time and resources required.

Table 2-2 Summary of Needed Technical Work (ECCS Acceptance Criteria)

Technical Work	Calendar Time to Perform Work	Resources Needed
• Research to explore post-quench ductility of high-burnup zircaloy cladding • Interpretation of test results and formulation of the technical basis for a regulatory guide	4 months[1]	0.2 FTE $0K[2]

[1] Due to the need for input from ongoing research at Argonne National Laboratory, technical work to support these modifications can not be completed before the end of calendar year 2001.

[2] Funding for the research at Argonne National Laboratory is already in the budget, so additional funding would not be needed.

3. Description of Risk-Informed Voluntary Alternative to 50.46 and Associated Technical Work

This section provides a description of the proposed voluntary alternative to 10 CFR 50.46, Appendix K and GDC 35 which addresses the ECCS functional reliability. Each section provides a description of the proposed technical requirement and a summary of the technical work that is needed to support the rulemaking effort.

3.1 Proposed Change

Current requirements postulate loss of offsite power (LOOP) and single failures in design-basis LOCA analyses in order to assure that the ECCS function can be achieved reliably. Risk analysis techniques provide a more realistic approach for assessing and ensuring reliability.

Two options to the existing LOOP/single-failure assumptions are proposed:

- The first option would replace the simultaneous LOCA/LOOP requirement and the single failure criterion now used to accomplish system reliability with deterministic system reliability requirements based on risk information. As such, it is anticipated that LOCAs in different frequency intervals would have different system reliability requirements applied to them. The different LOCA frequency intervals and associated system reliability requirements would be established by the NRC based on generic information for the industry as a whole, or by plant type, and once implemented, licensees would be free to choose this alternative without any NRC review or approval. The ECCS reliability information could be used to establish generic ECCS requirements in several different areas. First, for specific LOCA sizes where the LOOP contribution to ECCS unreliability is below a specified value, the thermal-hydraulic analyses required in 50.46 for these LOCA sizes would not require the assumption that offsite power is available. Second, the ECCS reliability information could be used to potentially specify the minimum features of the ECCS design required for different LOCA sizes (e.g., an ECCS design requirement that only one train of ECCS is required for LOCAs larger than a specified size). Based on these requirements, licensees could pursue modifications to operational limits (e.g., technical specifications) or, under some conditions, modifications to system design. Any generic changes to the ECCS requirements based on reliability arguments, however, would have to be based on the reliability of the ECCS for all accidents, not just LOCAs, unless there is an ECCS component that is only required for LOCAs (e.g., the accumulators). Third, the ECCS reliability information could also resolve the single passive failure issue by allowing any identified single passive failures in the ECCS design as long as the cumulative contribution of these failures to ECCS reliability is small.

- The second option would allow each licensee to use plant-specific information to establish their own LOCA frequency intervals and ECCS reliability requirements, commensurate with the LOCA frequency. This information could include plant-specific PFM calculations to establish LOCA frequencies and ECCS reliability evaluations that reflect unique plant features and plant-specific operational data. In this alternative, the NRC would set an acceptable threshold value for the frequency of core damage associated with a specified set of LOCA initiators. The licensee would be required to meet these core damage thresholds values using plant-specific LOCA frequencies and ECCS reliability values, and due consideration of uncertainties. Suppose, for example, that the threshold for all pipe-break-initiated LOCAs was set at 10^{-5} per reactor year. Let CDF_{LOCA} denote the corresponding

LOCA-initiated core damage frequency (CDF) estimate for a particular plant. To exercise the alternative would require that CDF_{LOCA} be less than the NRC-specified threshold value. If this were true, licensees could then use reliability analysis techniques to reduce the LOOP requirement for some LOCA sizes in the thermal-hydraulic calculations required by 50.46. In addition, the plant-specific analysis could be used to modify operational limits (e.g., technical specifications) or, under some conditions, to modify system design. Any plant-specific changes to the ECCS requirements based on reliability arguments, however, would have to be based on the reliability of the ECCS for all accidents, not just LOCAs, unless there is an ECCS component that is only required for LOCAs (e.g., the accumulators).

3.2 Needed Technical Work

The needed technical work to support rulemaking for the proposed changes associated with ECCS functional reliability involves the following three principal areas:

- LOCA frequencies
- Conditional probability of LOOP given a LOCA
- ECCS risk/reliability analyses

The required technical work associated with each of these areas is briefly described in the following sections. Section 3.2.4 summarizes this information and provides estimates of the time and resources required.

3.2.1 LOCA Frequencies

Technical analysis will be required to identify frequencies of LOCA break sizes. One important frequency value could be associated with the break size that represents the transition point to LBLOCAs often used in probabilistic risk assessments (PRAs) (e.g., >6 inches for PWRs). One possibility for obtaining the frequency of this transition point and the frequency of other LOCAs might involve a review of current LOCA frequencies used in PRAs, and consideration of pipe break frequency data and insights, to determine an appropriate frequency based on engineering judgement. Another possibility might involve an update of the NUREG/CR-5750 analysis of pipe-break LOCA frequencies, to account for more recent operating experience. For this possibility, a less subjective analysis of uncertainties in the frequency estimates for LOCAs may be required to assure that suitably conservative estimates are used in implementing those alternatives requiring analyses of LOCA initiator frequencies or LOCA CDF contributions.

Alternatively, probabilistic fracture mechanics (PFM) and a review of service history data could be used to identify the pipe diameter corresponding to a desired break frequency; however, this would be both costly and time-consuming. The technical work to support this option would be similar in scope to the work required for redefinition of the LBLOCA (see Section 4.2).

3.2.2 Conditional Probability of LOOP Given a LOCA

Technical analysis will be required to support the development of regulatory guidance that would specify acceptable methods and assumptions for quantifying the conditional probability of LOOP given a LOCA, and for identifying those plant-specific features that tend to decrease the likelihood of LOOP given an LOCA (e.g., plant capability to communicate with transmission system operators). All potential causes of LOOP given a LOCA would be considered. Both consequential and random LOOP events would be considered. Both prompt and delayed LOOP events would be considered. Earlier analyses of LOCA-LOOP frequencies [Ref. 2], which were performed as part of the resolution of Generic Safety Issue (GSI)-171, would be updated and upgraded to account for subsequent insights, operating experience, and the impact of electric power deregulation on grid stability. Prior analyses of automatic bus transfers would also be reexamined [Ref. 3]. These analyses would identify and/or confirm features that tend to decrease the likelihood of LOOP given a LOCA and compile relevant data. In particular, plant capabilities to communicate with transmission system operators, as mentioned above, and the availability of data on electric grid stability would be evaluated. The regulatory guidance may also specify methods for monitoring (e.g., electric grid stability) to assure the preceding analyses remain valid.

3.2.3 ECCS Risk/Reliability Analyses

Technical analysis will be required to support development of regulatory guidance that would specify acceptable methods and assumptions for performing LOCA CDF and ECCS reliability analyses for those alternatives requiring such analyses. In addition, appropriate reliability and CDF threshold values would have to be selected. Some specific issues that would need to be addressed include:

a. The ECCS success criteria in the CDF calculation are based on a CDF definition whereas the ECCS performance acceptance criteria are based on peak cladding temperature (PCT) and cladding oxidation (i.e., cladding ductility). Consideration will have to be given as to whether the definition of CDF should be the same as the ECCS acceptance criteria when utilizing this risk-informed alternative.

b. The proposed risk-informed alternative only considers pipe break LOCA initiators in this evaluation consistent with the scope of the current 10 CFR 50.46. However, to be completely risk-informed, all LOCAs (including stuck-open relief valves and seal LOCAs) should be considered when evaluating ECCS acceptability.

c. A determination will need to be made as to what are appropriate LOCA frequencies to use in these evaluations. Specifically, it will have to be decided whether operational data estimates such as in NUREG/CR-5750 are sufficient, or whether PFM analyses are required.

d. When evaluating the ECCS reliability, some issues not currently modeled in PRAs may need to be addressed. These include dynamic effects such as asymmetrical loads, pipe whip, and jet impingement; delayed LOOP and degraded voltage issues; and sump plugging. Guidance for treating these issues probabilistically could be required. In addition, guidance might be required for addressing some potential ECCS risk issues that have been inconsistently treated in past PRAs, as discussed in Section 4.4.2 of Attachment 1.

e. Since the use of mean CDFs could be allowed in the alternative, guidance with respect to the scope of needed uncertainty analyses might need to be developed.

f. The thermal-hydraulic (T-H) calculations cover a limited period of time during blowdown and recovery. The ECCS for PWRs would be in the injection mode. Thus, if the reliability evaluations are to be used for making decisions on the T-H conditions, the reliability calculations should be limited to evaluating the injection mode of ECCS and not the recirculation mode.

g. If the ECCS recirculation mode reliability is to be assessed, that information could be used in some fashion to address the long-term cooling requirement in 10 CFR 50.46. Currently, no calculations (PRA or T-H) are used to evaluate compliance with this requirement. Sump plugging is clearly a recirculation mode issue that could be addressed in a risk-informed 10 CFR 50.46 based on a reliability calculation. An appropriate mission time would need to be identified for use in this recirculation evaluation (e.g., 30 days).

h. When evaluating the LOCA CDF, it is currently envisioned that only ECCS systems would be credited. However, further consideration may need to be given to expanding the reliability calculations to include non-ECCS systems. The implications of this would need to be identified (e.g., would a non-ECCS system credited in the CDF evaluation have to be subjected to an Appendix K T-H calculation).

Demonstration analyses for selected plants may be helpful in developing the needed regulatory guidance.

3.2.4 Summary

A summary of the needed technical work to support rulemaking for the proposed changes associated with ECCS functional reliability is provided in Table 3-1. This table also includes estimates of the time and resources required.

Table 3-1 Summary of Needed Technical Work (ECCS Functional Reliability)

Technical Work	Calendar Time to Perform Work	Resources Needed
• Determine appropriate LOCA frequencies for use in LOCA CDF and ECCS reliability analyses • Identify features that tend to decrease the likelihood of LOOP following a LOCA, and develop guidance for estimating plant-specific probability of LOOP given a LOCA • Develop guidance for performing LOCA CDF and ECCS reliability analyses • Determine appropriate reliability and CDF threshold values	6-9 months	0.5 FTE $350K

4. Description of Additional Changes to 50.46 (Spectrum of Breaks) and Associated Technical Work

This section provides a description of the proposed additional changes to 10 CFR 50.46, Appendix K and GDC 35 which addresses the spectrum of breaks. Each section provides a description of the proposed technical requirement and a summary of the technical work that is needed to support the feasibility study.

4.1 Proposed Change

In 10 CFR Part 50, there are two places where the rules define loss-of-coolant accidents as hypothetical or postulated *"accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system"* (10 CFR 50.46(c) and 10 CFR Part 50, Appendix A, "Definitions and Explanations" -- note, in the definition in Appendix A, the words "in pipes" are not included). In addition, 10 CFR Part 50, Appendix K, paragraph (I)(C)(1), states that *"In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system."* The industry believes that the top priority in risk-informing 50.46 is redefining the maximum break size for the LBLOCA; i.e., defining a new maximum LOCA break size above which breaks would no longer need to be included in the plant design basis. The feasibility assessment for this effort has not yet been completed and recommendations are not proposed by the staff in the near term. However, if in the future, the industry, or an applicant, can successfully justify that a given set of LOCAs has a collective frequency below a specified threshold value, for example, 10^{-5}/yr, the staff would consider reduction of regulatory requirements to be commensurate with this frequency (e.g., allow best-estimate calculations without the need for uncertainty propagation, relax reporting requirements, or relax technical specifications). The staff has agreed to engage with the public in a series of meetings to further define and refine the feasibility effort for redefining the LBLOCA. The underlying concerns of degradation mechanisms, materials properties, and uncertainty analyses will be addressed, as well as issues of resources required to accomplish this work.

4.2 Needed Technical Work

In the longer term, the staff would continue to work with industry to establish the feasibility of LBLOCA redefinition, and develop regulatory guidance for plant-specific submittals (or, possibly, submittals by plant or reactor type). Technical work and resources to support redefinition of the LBLOCA are summarized in Table 4-1. Included in this work is development of an analytical approach, as described in Appendix A to this attachment. The results of this work will be conveyed to the public and industry through a series of meetings. Then, existing computer codes must be reviewed, adapted, undergo quality assurance checks, and be benchmarked in order to be applicable to redefinition of the LBLOCA. Existing codes are deficient for this purpose. The benchmarking would involve the use of sample plant evaluations or sample pipe system evaluations. These results would be compared to the results of other codes to ensure that any discrepancies are fully understood. A key result of this effort will be the identification of break frequency to a corresponding pipe diameter. This function cannot be extrapolated based upon a few data points, rather frequencies must be determined for each pipe diameter, since pipes of varying diameter will be exposed to different stresses and degradation mechanisms and thus result in different break frequencies.

Table 4-1 Summary of Needed Technical Work (Spectrum of Breaks)

Technical Work	Calendar Time to Perform Work	Resources Needed
Establish Feasibility and Support Development of Regulatory Guidance: • Develop analytic approach (including series of public meetings to convey technical requirements) • Review of existing computer codes • Updating/developing computer codes • Sample plant or pipe system evaluations • Code comparisons	2-3 years	2.4 FTE $1.2M

5. REFERENCES

1. USAEC Opinion of the Commission in the Matter of “Rulemaking Hearing, Acceptance Criteria for ECCS for Light Water Cooled Nuclear Power Reactors,” CLI-73-39, December 28, 1973.

2. G. Martinez-Guridi, P. K. Samanta, T. L. Chou, and J. W. Chang, "Evaluation of LOCA With Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios," NUREG/CR-6538, July 1997.

3. S. Mazumdar, “Engineering Evaluation Report: Operational Experience on Bus Transfer,” AEOD/E90-05, June 1990.

APPENDIX A

TECHNICAL CONSIDERATIONS FOR REDEFINING THE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT

APPENDIX A

Introduction

Leak-before-break (LBB) has been historically applied in many industries. In the nuclear industry, it has been applied to nuclear piping, primarily PWR Class 1 or primary piping systems. These analyses have been conducted since the draft standard review plan (SRP 3.6.3) for LBB was issued in 1983. The NRC's LBB procedure is a deterministic two-step approach with certain screening criterion. The screening criterion essentially requires that no active degradation mechanisms that can cause long surface flaws exist in the pipe system of interest, and that there are not any unquantifiable high stresses (i.e., water hammer) in the pipe system. Hence, LBB has traditionally been applied with some experienced-based risk knowledge. NRC LBB Draft SRP 3.6.3 requires computational procedures for all pipe systems, including those with no detectable flaws and low stresses. The LBB computational procedure is meant to demonstrate sufficient flaw tolerance for the material being evaluated using existing leakage detection requirements from NRC Regulatory Guide 1.45.

In several ways the current NRC Draft SRP 3.6.3 LBB procedure is simpler than if a leakage calculation and fracture assessment was to be done for an actual service crack. If risk-based analyses are used to generically reduce large-break LOCA requirements for 10CFR50.46, then the LBB methodology needs to include all the complications of making an actual leakage assessment as well as consideration of unknown future degradation mechanisms. The following general methodology is a broad summary of the procedures that are believed necessary to make these calculations with a greater degree of confidence than is currently used for LBB.

The following inputs are necessary to make a generic risk-based assessment:

- Initial flaw distributions,
- Pipe-system boundary conditions and postulated flaw locations,
- Degradation mechanisms (flaw growth evaluations, as well as crack morphology parameters for leak-rate calculations),
- Normal operating and transient loads,
- Material response, i.e., strength, toughness, including aging effects or loading-rate effects on material properties,
- Leak-rate estimations,
- In-service inspection, and
- Uncertainty analyses including all the above variables.

Some preliminary recommendations of these different input variables are given below.

Initial Flaw Distribution

An initial flaw distribution is used for life calculation in structures. This has been done for reactor pressure vessels for pressurized thermal shock. Initial flaw distributions could be determined from inspection results and validation of inspection results, i.e., cutting apart pipe weld, and expert processes and computer codes, i.e. PRODIGAL. Work in the PISC program (an international Program for the Inspection of Steel Components) may be helpful. Additionally there are requirements for pre-service flaw sizes in ASME Section XI Article IWB-3500. Records of acceptable flaw sizes from pre-service inspections would be helpful if they are available. The probability of multiple flaws in a weld should also be included.

Degradation Mechanisms

Degradation mechanisms are important from several considerations. First, will there be some degradation mechanism that will cause the initial flaws to grow. Secondly, and perhaps more important, is whether the mechanism of interest will result in long surface flaws as occurred in intergranular stress corrosion cracking (IGSCC) flaws in boiling water reactor (BWR) plants in the late 1960 and 1970's. Thirdly, the degradation mechanism will also affect the surface roughness and straightness of the flow path for leakage considerations. LBB analyses assuming smooth fatigue cracks would give shorter crack lengths for a given leak rate than IGSCC or primary water stress corrosion cracking (PWSCC) cracks with rougher surfaces that might actually occur in service.

In regards to flaw length development, the degradation mechanism could force either short or long surface flaws to develop. For instance, intergranular stress-corrosion crack growth is dominated by girth weld residual stress fields in the heat-affected zone of the base metal adjacent to the weld. These stress fields can be high and relatively uniform around the circumference. The service stresses may not be non-uniform enough around the circumference to make the surface flaws grow only over a short length. Thermal fatigue loads or thermal striping can cause cyclic loads over either large areas or very short lengths, respectively, and hence have different significance on LBB behavior.

As previously described, PWSCC cracking has occurred in small-diameter instrumentation lines, steam generator tubes, control-rod head-penetration tubes, and more recently in bimetallic lines of the large-diameter primary piping in PWRs. The cracking in the control-rod head tubes was first axial with little structural concern, but more recently some circumferential cracks have occurred that are of concern. The bimetallic weld metal cracks in PWR large-diameter piping were mainly axial cracks, but the potential for circumferential cracks forming needs to be assessed for continued LBB acceptability.

Generic long-term acceptance of LBB behavior for reducing LBLOCA requirements needs to consider these types of mechanisms, as well as the possibility of long-surface flaw degradation producing mechanisms.

Normal Operating and Transient Loads

In typical LBB analyses, the normal operating loads need to be considered for leakage detection capabilities. Normal operating loads are also needed for flaw growth analyses in probabilistic evaluations. The normal operating loads need to consider:

- pressure,
- dead-weight loads,
- thermal expansion,
- residual stresses,
- cyclic thermal loads, and
- fabrication stresses.

The first of these three stress contributors are included in typical design reports. Residual stresses are important for crack initiation and growth considerations, and their effect on the crack opening for leakage rates should also be addressed. Cyclic thermal loads such as thermal striping are not design-basis loads and need to be considered in probabilistic flaw growth analyses. Fabrication stresses are difficult to quantify, but should be evaluated. It is well known that in pipe system repairs, the pipes may spring apart when cut. Some determination of the equivalent stresses from such realistic behavior should be included. Another non-design stress is the effect of weld repairs (i.e., weld overlays) on stresses in other locations of the pipe system. These stresses should also be included in the analysis.

Hoop stresses are the controlling stress for axial flaws. For circumferential flaws, leakage and fracture are controlled by longitudinal membrane (from pressure) and bending stresses, but torsional stresses should also be included. Methods to determine equivalent bending stresses for combined bending and torsional stresses to determine leakage and fracture behavior exist.

Transient loads are considered to typically be the safe-shutdown earthquake (SSE) loads. This is a design load that occurs with some probability of occurrence. However, other greater and lesser earthquake loads can occur with different probabilities and should be included in a complete probabilistic analysis. In some pipe systems, the thermal transients during start-up or shutdown can result in higher stresses than the SSE loads. Such thermal transient loads should be included in the crack growth and fracture analyses. It is highly unlikely that an SSE event would occur during a transient thermal load, but if the thermal loads occur for a longer time period, then the combined load probability should be included.

Pipe-System Boundary Conditions and Postulated Flaw Locations

The pipe dimensions, pipe supports, and other pipe boundary conditions need to be specified. The location of a postulated crack in the pipe system determines the stresses, but an often overlooked aspect is that the pipe-system boundary conditions can also affect the leakage and fracture calculations from the viewpoint of how the pipe rotation is restricted by the rest of the pipe system. Typical LBB leakage analyses assume the ends of the pipe are free to rotate like an end-capped vessel. Restraint of this rotation will make the leakage size flaw larger and the same transient loads will be more detrimental to the structural integrity.

In LBB analyses, the location with the worst material properties and highest stresses is typically considered. Probabilistically, there could be a combination of material properties, lower normal operating stresses and slightly lower SSE stresses that might be more detrimental to fracture behavior than the location with the highest SSE stresses.

Flaw orientation is also an important consideration for actual flaw assessments. In hypothetical flaw analyses for LBB, the flaws are typically circumferential flaws in straight-pipe girth welds. Axial flaws in nuclear piping are rare, except for erosion-corrosion flaws in Class 2 or 3 piping which are not of interest here. For circumferential flaws, existing LBB analyses assume that the flaw is centered on the bending plane for normal operating stresses and the bending plane under the transient stresses. Using the location of the flaw being centered on the bending plane for transient stresses is conservative for the fracture analyses, but using the center of the bending plane for the leakage calculations is the least conservative flaw location around the pipe circumference. Realistically, the normal operating stress bending plane and the transient bending planes are probably different. More importantly, fabrication flaws can occur anywhere around the pipe circumference and may not be located on either bending plane. Some analyses exist for consideration of the effects of off-centered crack for leakage and fracture analyses, and should be incorporated into a generalized probabilistic approach.

The flaws could be located in the weld metal, base metal, or heat-affected zones (HAZ). Typical LBB analyses use the lower of the weld metal or base metal properties. However, cracks in stainless steel weld tests frequently grow and follow the fusion line of the weld. Consideration of the toughness in the HAZ or fusion lines should be included in a probabilistic analysis.

The occurrence of flaws in elbows and other pipe fittings than just straight pipe should also be considered.

<u>Material Response</u>

Some of the key material response factors that should be considered in any risk-based analyses are:

- the strength of the material,
- fatigue/environmental crack initiation properties,
- fatigue crack growth rate including environmental effects, and
- fracture toughness.

The strength of the material changes with temperature, so that the operating temperature needs to be considered. Code strength or actual strength values could be used in deterministic analyses, but statistical variability of the material strength (yield, ultimate, and strain-hardening exponent) could be employed in a probabilistic analysis.

Fatigue initiation life with environmental effects (i.e., ?K_{th} or K_{Iscc}) could use lower-bound values or a statistical distribution if enough data exists. These values are for the temperate, water chemistry of interest and vary by material. Even within a class of materials, the effect of certain elements, i.e., sulphur in ferritic steels, can change the material sensitivity to environmental effects. Loading rates in service and hold times can also effect the environmental crack growth rates.

The material fracture toughness values can differ significantly between the base metal and weld metal. Wrought stainless steels usually have a very high toughness, but that toughness may be lower if there is a higher amount in impurities such as sulphur and phosphorous. The type of weld procedure can greatly effect the toughness and needs to be considered. For stainless steels, SMAW and SAW welds statistically have the same toughness which is significantly lower than TIG welds. There is also some evidence that the toughness in the fusion line of austenitic welds can be lower than the weld metal, so that fusion line toughness and heat-affected zone (HAZ) toughness values should also be included in a probabilistic distribution of properties for the degradation mechanism of interest.

There are many factors that can also affect the fracture toughness, such as

- thermal aging of duplex stainless steel materials,
- carbon depletion in some bimetallic welds,
- dynamic strain aging in ferritic steels and weld, and

- cyclic loading.

Thermal aging of cast stainless steels (particularly those with higher ferrite content) has been well documented and can cause the base metal toughness of pipe or elbows to be much lower than wrought stainless steel or even lower than the weld metal toughness. Stainless steel welds can have duplex structures. Thermal aging effect on lowering the toughness of stainless steel welds is somewhat known, but has not been used in LBB evaluations to date.

Bimetallic welds can have a carbon depletion problem in the ferritic material heat-affected zone that will lower the toughness in that region. This would occur in older nuclear plants where the bimetallic weld were entirely made of stainless steel weld metal with no buttering of the ferritic steels using Inconel weld metal.

Since seismic loads are both dynamic and cyclic, effects from these loading conditions should be considered. Dynamic strain aging is an effect that can raise the ultimate strength, reduce the percent elongation in tensile tests, and lower the fracture toughness of carbon steels under certain conditions. Dynamic strain aging is a dislocation pinning phenomena which is sensitive to temperature and loading rate. The dislocation pinning can be by free atomic nitrogen or carbon atoms and occurs at temperatures between 300F and 650F for ferritic steels and their welds. This temperature range is in the operating temperature range of primary nuclear piping systems. In more recent years the effects of loading rate on the toughness of ferritic steels has been determined to be important for ferritic steels due to this dynamic strain aging effect. The loading rate to be used in testing should correspond to reaching crack initiation in the fracture test in the time that corresponds to one-quarter of the period of the first natural frequency of the pipe system of interest.

There has also been some results showing that cyclic loads during ductile tearing can also lower the fracture resistance. This is much harder to quantify, but generally if the ratio of the minimum to maximum stresses (in the presence of a crack) is more negative than -0.5 then cyclic loads could be detrimental to the material fracture resistance.

<u>Leak-Rate Estimation</u>

Leak-rate evaluations are made for several purposes. First, in LBB analyses, the leak-rate is used as a crack detection method. Secondly, the residual crack opening after a transient event can determine if the crack resulted in a small-break or large-break opening. Finally the rate of crack opening during a large-break determines the thermal-hydraulic loads on the reactor core supports, as well as pipe whip and jet impingement forces.

The NRC LBB methodology for nuclear piping involves a two-step analysis procedure. The first is to determine the crack size that corresponds to the detectable leak rate at normal operating conditions with some safety factor on leak-detection capability. That crack must be stable under transient loads with an additional safety factor on the crack length.

There are a number of leak-rate related factors to consider in making such an assessment that may be pertinent in a risk-based analysis. These factors include:

- crack morphology parameters (i.e., roughness, number of turns in the crack path, general straightness of the crack path through the thickness),

- effect of pipe-system boundary conditions (see earlier discussion) on restraining the crack-opening displacement for a cracked pipe under internal pressure,
- effect of weld residual stresses on the crack-opening displacements,
- effect of crack-face pressure,
- effect of off-center cracks, and
- potential of particulate plugging with tight cracks.

The choice of the crack morphology parameters can have a major role in estimating the detectable leakage crack size, and thus the probability of pipe break occurring. Analysis have shown that an LBB applicant's choice for crack morphology parameters resulted in a predicted leakage size flaw about half the crack size as when the statistical values of surface roughness and number of turns from service cracks were used. These crack morphology parameters are dependant on the degradation mechanism, and care should be undertaken in selecting the proper values as well as validated leak-rate computer codes. Statistical variability of the crack morphology parameters has been defined in NUREG/CR-6004 for several degradation mechanisms.

The restraint of pressure-induced bending has been determined to be an important aspect in some LBB analyses to date. This comes about from the crack-opening analyses in typical LBB applications assuming that the ends of the pipe are free to rotate when there is a circumferential crack in a pipe under internal pressure. The pipe-system geometry and boundary conditions (from pipe supports, nozzles, elbows, etc.) will restrain that pipe rotation and hence have a lower crack-opening displacement. This means that for the same leak rate, the crack length would be longer when considering the pipe-system boundary conditions on the crack-opening displacement. Since the effect of restraint of pressure-induced bending is strongly a function of crack size, this effect will have a bigger impact on small-diameter pipe than large diameter pipe, except perhaps for steam lines where larger cracks are required to get the same leakage as in subcooled-water lines.

From an LBB perspective, weld residual stresses are more important for "thin-wall" pipe than they are for "thick wall" pipe, and since thin-wall pipes are more closely associated with smaller diameter pipes, this effect will also be more important in smaller-diameter pipes. Since the effect of weld residual stresses in thin-wall pipe is to rotate the crack faces such that flow through the crack is choked off, ignoring this effect could cause an under prediction of the pipe break frequency for the smaller-diameter pipes.

The effect of crack-face pressures is to open the crack more so that leakage detection is easier for shorter cracks. This effect is seldom used in LBB analysis.

As noted in the section on crack location, having a circumferential crack off-centered from bending plane will reduce the crack opening and result in having a longer crack for the same leak rate. Some analyses exist to account for this behavior.

Leak-rate codes that give similar accuracies are the PICEP and SQUIRT codes. The uncertainty in these models have been established by comparisons to experimental data. Accuracies to plus or minus a factor of 2 is typical. Both of these leak-rate codes can have problems with the friction factor partes of their analyses, that is accounted for by using global roughness values verus local roughness parameters as explained in NUREG/CR-6004. Plugging by particulates with tight cracks may also be a concern.

Leak rate analyses can be used to quantify thrust loads imparted. Obviously, this sort of analysis is totally different than that invoked in a LBB assessment. This type of analysis will involve single-phase flow through a large break where such factors as the crack-morphology parameters are of little concern. Furthermore, before using any such analyses, it would be necessary to provide some level of validation. Further note that some leak-rate codes, such as SQUIRT, are not appropriate in this regime since they were written for two-phase flow cases only.

In-Service Inspection

In-service inspection (ISI) capabilities have increased significantly since the building of the initial nuclear power plants. Nevertheless, there are certain material/flaw combinations that are difficult to detect or size. The probability of detection needs to be incorporated in the risk-based analyses.

Typical detection of leaking cracks has frequently been determined by visual observations, not by sophisticated leak-detection equipment. Water puddles on the floor, or piles of boric acid powder in areas have been detection sources in many early cases of a particular degradation mechanism.

Risk-based inspection techniques have told operators to look at historical cracking locations. This is a necessary but not complete approach. Such methods may have told operators to minimize or eliminate inspections from other areas where cracks have not been seen in the past, but may be the source of new cracking mechanisms. The occurrence of PWSCC in hot legs is one such example.

Nevertheless, the ability to nondestructively size flaws is a highly desirable capability. Even if flaw depths cannot be accurately determined, the determination of flaw lengths is essential in augmenting LBB behavior for a degradation mechanism can produce long-surface flaws.

The assistance of ISI in improving LBB behavior should be considered in a risk-based LBB analysis, but the detrimental aspect of focusing resources away for locations where future flaw degradation mechanisms may develop needs to also be included.

Uncertainty Analyses

Several probabilistic models exist for determining the risk and uncertainties in the risk evaluations. One of the earlier ones was the PRAISE code, which was modified for the Westinghouse SRRA (Structural Reliability and Risk Assessment) code. These codes have made improvement since the early days of the PRAISE code from the days of the resolution of Generic Issue A2 on asymmetric blow down loads on older nuclear power plants. However, they are lacking in many of the deterministic leak-rate and fracture considerations in the NUREG/CR-6004 procedure as well as many of the above considerations.

A more complete leak-rate and fracture analysis is included in NUREG/CR-6004 for probabilistic LBB analyses, but even since the NUREG/CR-6004 report in 1995, many additional deterministic improvements have been made. The NUREG/CR-6004 probabilistic analysis does not include calculations of the probability of crack initiation and crack growth, but determines the conditional probability of a crack occurring with a given leak rate and no imposed safety factor for determination of the failure probability of an SSE event. Although useful in showing that the leakage size flaw and normal operating stresses are more important than the SSE loads in determining failure probabilities, the NUREG/CR-6004 study was only a sensitivity analysis that was not carried out far enough.

The PRAISE, SRRA, and NUREG/CR-6004 analyses are lacking in determining the failure probabilities with the deterministic analyses that currently exist as well as many of the considerations discussed above.

Other Failures

One aspect about the double-ended pipe fracture assumptions for ECCS sizing, which is not directly addressed, is that this loss-of-coolant sizing approach not only accounts for hypothetical cracks occurring in the piping systems, but also accounts for many other potential failure causes that are not related to cracks occurring in pipes. The LBLOCA pipe break analysis is a surrogate for these mechanisms, as well as the more frequently analyzed hypothetical pipe breaks. Some examples of these other failure modes that are covered by the LBLOCA analysis include:

- failure of the bolting for steam generator manways,
- failure of bolts on valve bonnets for LOOP stop valves,
- multiple failure sources, i.e., control rod drive mechanisms (CRDMs), and
- indirect sources.

During the initial nuclear piping LBB work at Lawrence Livermore for the evaluation of Generic Issue A2 (on asymmetric blow-down loads on older nuclear plants without pipe restraints at the vessel nozzles), some consideration was mode for indirect causes leading to pipe breaks. One such cause was the dropping of a large object from an overhead crane. The movement of heavy equipment in the containment building while the plant is under operation should have careful consideration. Similarly, bolt degradation criteria and protection from new bolting failure mechanisms should also be considered.

As noted earlier in this Attachment, the current risk-based models are good for near-term extrapolation of failure frequencies. However, if some new degradation mechanism reaches its incubation time, then the probabilities calculated from those risk analyses are invalid. Hence, these future degradation mechanisms may be considered to be precursors to future failure modes. By reducing the LBLOCA requirements, the protection against these other failure modes is reduced.